International Journal of Multiphase Flow 83 (2016) 51–61

Contents lists available at ScienceDirect

International Journal of Multiphase Flow

journal homepage: www.elsevier.com/locate/ijmultiphaseflow

# Numerical investigation of the boiling crisis for helical cruciform-shaped rods at high pressures

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ARTICLE INFO

*Article history:*
Received 22 October 2015
Revised 23 February 2016
Accepted 12 March 2016
Available online 18 March 2016

*Keywords:*
Boiling crisis
Multiphase flow
Nuclear power
CFD

ABSTRACT

The limiting factor in heat removal in engineering systems involving liquid heat transfer is the boiling crisis. The boiling crisis is especially important in the field of nuclear engineering, as it is one of the main limits on the power extracted per unit volume for both Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR). The Helical Cruciform Fuel (HCF) bundle has been proposed to increase the power density in PWRs and BWRs through enhancement of the heat transfer area-to-volume ratio and mixing of the flow compared to the traditional cylindrical fuel rod bundle geometries. In order to explore the HCF improved performance limits, computational multiphase flow dynamics is pursued here, as the experimental testing of the conditions of interest in BWR and PWR is costly. The boiling crisis at low qualities is characterized by Departure from Nucleate Boiling (DNB) and at high qualities is understood to be the result of liquid film dryout. The two-phase Eulerian approach was used to develop a robust baseline framework for prediction of DNB under high pressures that predicts the trends in boiling crisis for engineering scale systems. Such numerical technique agreed with the trends observed in the collected experimental data related to HCF type geometry.



## Introduction

The boiling crisis or Critical Heat Flux (CHF) is a thermal limit in heat transfer for which a heated wall loses its ability to effectively transfer heat to the coolant. The accurate prediction of such thermal limit is critical in the design of heat transfer surfaces in many engineering applications including fossil and nuclear power plants and electronics cooling. Current practice engineering approach to quantify this limit depends on empirical correlations based on experimental data. In many applications, such as high-pressure conditions of nuclear fuel rod bundles, performing such experiments is very costly. Recently, Computational Fluid Dynamics (CFD) simulation has been more extensively used for multiphase systems such as boiling. Recent work shows promise in prediction of boiling heat transfer using the CFD framework, such as Roy et al. (2002) and Yun et al. (2012) in subcooled boiling and by OECD benchmark in saturated flow boiling. Though, such work typically does not show extensive benchmarking vs. many databases implying lack of model robustness.

Typically, in order to increase the power output of a system before reaching the boiling crisis, enhancement in heat transfer coefficient through use of fins, twisted tapes and more recently nanofluids and surface modifications is pursued. The latter two approaches are still under investigations, as nano-engineered surfaces and prediction of nano-particles are still not plausible through the current state of fluid dynamics. For nuclear fuel applications, the Helical Cruciform Fuel (HCF) geometry (Conboy, 2010), shown in Fig. 1, uses the strategy of fins to increase the heat transfer area and also the twisted tapes approach to increase the swirl and intra-bundle mixing of the flow to increase the CHF value compared to the traditional cylindrical fuel rod bundle geometries. The cylindrical rod bundles are used in the two most common nuclear power reactors in the world: the Pressurized Water Reactor (PWR) and the Boiling Water Reactor (BWR). In a typical PWR or BWR, the nuclear fuel is encapsulated in a metallic cladding in a cylindrical geometry, where the heat generated in the fuel is conducted through the metallic layer and removed by the flowing coolant. The coolant flows upward between the cylindrical rods and the magnitude of heat transfer to the coolant is limited by the boiling crisis. In both PWRs and BWRs, cylindrical rods are held vertically with spacer grids to avoid contact and to minimize flow induced vibrations. However, the HCF rods are in contact at every 1/4th (e.g. 90 degree) twist pitch and do not require spacer grids as they support each other. The spacer girds provide the most challenge in computational modeling of fuel bundles, which has made licensing of new fuel rely on experimental tests. In addition, the grids are a source of additional pressure drop and the number one cause of fuel failures in PWRs in the World (Inozemtsev et al., 2013).

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http://dx.doi.org/10.1016/j.ijmultiphaseflow.2016.03.014
0301-9322/

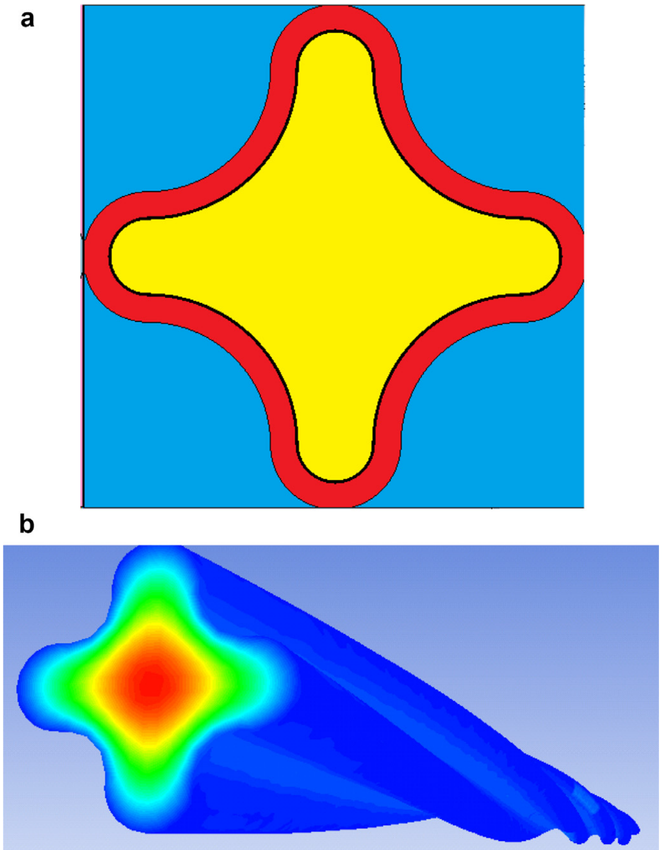


**Fig. 1.** The drawing of the (a) HCF design cross-section and (b) 3D temperature distribution simulation.

In a PWR, where the coolant remains subcooled throughout the core, subcooled nucleate boiling could occur, especially in the USA operated power plants. The boiling crisis in a PWR is typically called the Departure from Nucleate Boiling (DNB) and empirical correlations are used to predict its magnitude. These correlations are functions of pressure, subcooling, heat flux, coolant mass flux, axial power shape and hydraulic diameter, as well as the spacer grid configuration. In BWRs, the boiling crisis is typically referred to as critical power limit, as it related to the total power that will lead to liquid film dryout on one or more of the heated rods. Most of the BWR core is cooled by saturated boiling with the annular flow regime present. In a BWR, the dryout limit (i.e. the critical power) is a function of pressure, subcooling, mass flux, hydraulic diameter, the ratio of the wetted to heated perimeter and total power transferred to the coolant (or the flow quality attained).

The fundamental difference between DNB and dryout is that DNB depends on the local heat flux, where rapid generation of bubbles could form a vapor film that insulates the heated wall surface at a particular location, while in a BWR, the dryout depends on the total heat transferred to the fluid (hence total steam flow) that leads to stripping or evaporation of the liquid film in the annular flow regime. The dryout boiling crisis is typically representative of what takes place in a steam generator in a variety of engineering applications. It is noted that DNB could also occur at high PWR pressures (15.5 MPa) at saturated boiling conditions. Due to this fundamental difference in flow regime, the same CFD models are not able to simulate both boiling crises.

So far, the two phase flow Eulerian approach for the CFD framework has proven most promising, both in terms of accuracy and computational resources required to perform mechanistic analysis of the boiling phenomena on an engineering scale. In the Eulerian framework, the liquid is modeled as a continuous medium, while the vapor is modeled as a dispersed phase. This is an ideal approach for the DNB phenomenon as the dryout has been typically simulated with mechanistic two dimensional multi-field models to separate the liquid film, vapor core and entrained liquid droplets fields that are all present in the annular flow regime. However, recent work by Hansch et al. (2012) has shown promising results by using a hybrid approach of Eulerian and field resolving methods to save computational time compared to the more expensive interface tracking methods.

In the last decade or so, the nuclear energy industry in the US has added over 5000 MWe to the PWRs and BWRs power output in form of plant uprates. Many of these uprates were approved due to a higher confidence in the measured and calculated operating parameters. Improving the boiling crisis limit prediction could give potential for further increase in output of existing plants as well as new nuclear power plants. Though, as of today, there are no operating facilities in the US for full rod bundle CHF testing for nuclear reactors. Most of the former CHF facilities were closed due to cost overruns.

The DNB and dryout rod bundle water experiments at pressures of the PWR (15.5 MPa) and BWRs (7.2 MPa), are extremely costly. One of about 200 PWR assemblies in a reactor with a typical peaking factor of 1.45 requires 27 MWth of energy at full scale, while operating at 15.5 MPa and 300 °C. Even at small scales such as a $4 \times 4$ array with fourth of the length of the rod at BWR pressures is estimated to cost ~$5 million to operate (Conboy, 2010). Thus, the use of computational analysis to predict CHF brings about cost savings. Therefore, any proposed new design, such as the above mentioned HCF design to improve the performance of nuclear engineering systems, needs to be carefully evaluated using computational methods to assure the presence of a good incentive for such costly experiments.

This paper expands on the previous work done on single phase flow experiment and CFD modeling (Shirvan and Kazimi, 2014). In this work, an overview of the applicability of existing experiments on HCF DNB and dryout performance as well as a qualitative picture of the HCF boiling crisis performance will be presented. Then, a fundamental computational study, in support of experimental data at high pressures, will be outlined using a robust 2 phase Eulerian model that could be used for PWR DNB predictions as well as other engineering systems at high pressures that require evaluation of the DNB margin.

## Background

The original design of HCF rods is based on a three petal version that was used in Russian ice breaker reactors. It was proposed for the hexagonal lattice VVER designs with metallic fuel (Bol'shakov et al., 2007). The four petal adopted for this study is more suitable for back-fitting the HCF rods in Western PWR and BWRs, as they use square lattice geometry. The previous studies by Conboy et al proposed the use of traditional oxide fuel, which is judged to be more acceptable from a safety perspective by the regulators. However, it will encounter manufacturing difficulties not addressed in this paper.

The HCF geometry in Fig. 1 has also a strong resemblance to a cylindrical pin with wire wraps, which is a relatively well known geometry in the nuclear engineering field. Therefore, the boiling experimental results for wire wrapped pins along with limited Russian three petal fuel design experiments are discussed in the following subsection. In addition, single-phase experiments as well as thermal hydraulic performance of HCF rods performed at MIT are discussed in this section.